NON-PUBLIC?: N
ACCESSION #: 9308110168
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Perry Nuclear Power Plant, Unit 1 PAGE: 1 OF 5

DOCKET NUMBER: 05000440

TITLE: Unexpected Reactor Recirculation Pump Fast to Slow Speed Downshift
EVENT DATE: 07/09/93 LER #: 93-015-00 REPORT DATE: 08/06/93

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Al Lambacher, Compliance Engineer TELEPHONE: (216) 259-3737 Extension 5520

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: AD COMPONENT: TE MANUFACTURER: R369
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On July 9, 1993, at 0537, with the reactor at 100 percent power, and both Reactor Recirculation Pumps in fast speed, both pumps unexpectedly downshifted automatically to slow speed. At 0544, Control Room operators manually scrammed the reactor from 52 percent power in accordance with plant operating procedures after the reactor entered the region of potential instability of the power to flow map.

It was concluded that the most probable cause of the Reactor Recirculation Pump speed downshift was the failure of both Reactor Recirculation Pump Suction Temperature Resistance Temperature Detectors (RTDs). Both RTDs were replaced and subsequently returned to the manufacturer for failure analysis. Any further corrective actions will be determined pending the results of this analysis. Additionally, a modification was made to provide a control room alarm should a RTD

failure recur. A time events analyzer has been installed to monitor the Reactor Recirculation Pump fast to slow speed downshift logic circuitry during the current Reactor startup.

END OF ABSTRACT

I. Introduction

On July 9, 1993 at 0537, both Reactor Recirculation Pumps unexpectedly downshifted from fast to slow speed. At the time of the Reactor Recirculation Pump speed downshift, the plant was in Operational Condition 1 at 100 percent rated thermal power. The reactor pressure vessel RPV! was at 1020 psig and saturated conditions. At 0544, Control Room operators manually scrammed the reactor after the reactor entered the region of potential instability of the power to flow map due to the reduction in core flow. At 0600 on July 9, 1993, the required non-emergency four-hour notification was made to the NRC pursuant to the requirements of 10CFR50.72(b)(2)(ii). This event is being reported under the requirements of 10CFR50.73(a)(2)(iv).

II. Event Description

On July 9, 1993, at 0537, with the reactor at 100 percent power, and both Reactor Recirculation System AD! Pumps P! in fast speed, both pumps unexpectedly downshifted automatically to slow speed. At this time, Reactor Recirculation Pump A(B) HI to LO Speed Transfer alarms SA! were annunciated in the control room. Control Room operators then entered Off Normal Instruction (ONI-C51), Unexplained Change in Reactor Power or Reactivity. At 0544, Control Room operators manually scrammed the reactor from 52 percent power in accordance with plant operating procedures after the reactor entered the region of potential instability of the power to flow map due to the reduction in core flow. There were no indications of power oscillations observed during the event.

As expected, the manual reactor scram caused reactor water level to decrease below Level 3 (177.7 inches above the top of active fuel) due to void collapse. Control room operators entered Plant Emergency Instruction (PEI-B13), Reactor Pressure Vessel Control. Water level was subsequently restored by the Feedwater System SJ!. The lowest vessel level reached was 157 inches above the top of active fuel. After water level was stabilized, Plant Emergency Instruction, PEI-B13, was exited at 0617. Plant shutdown was then continued in accordance with normal operating instructions.

At 0546, Control Room operators manually tripped the Main Turbine TRB!, after the turbine did not automatically trip to prevent reverse power, as the operators had expected. The manual turbine trip occurred with reactor pressure at approximately 850 pounds per square inch (gauge) and the Main Generator TG! indicating approximately 0 megawatts electric. An initial investigation identified that reverse power conditions were not present to initiate the trip logic, prior to the manual turbine trip. It was also determined, however, that the Main Turbine Control Valves PCV! did not stay closed. Subsequent investigations identified that additional adjustments to the Steam Bypass and Pressure Regulation System JI! logic are necessary.

At 0550, the Combustible Gas Control Hydrogen Analyzers BB! were started, as prescribed in PEI-B13. After the Hydrogen Analyzer startup was completed, Hydrogen Analyzer "A" indicated between 1.5 and 2.0 percent hydrogen concentration, while the "B" Analyzer read 0 percent concentration. The high reading on the "A" Hydrogen Analyzer was attributed to instrument drift and the Analyzer was declared inoperable. The "A" Hydrogen Analyzer was subsequently re-calibrated and returned to service.

At approximately 0625, Control Room operators noticed, after resetting the reactor scram logic, that the Scram Discharge Volume (SDV) first drain and vent valves V! did not re-open as required, until approximately fourteen minutes after the scram was reset. During subsequent troubleshooting it was determined that the slow opening times were due to excessive leakage past an air supply/exhaust valve V! common to both valves. The air supply/exhaust valve was replaced and will be returned to the vendor for failure analysis.

III. Cause Analysis

The initial investigation of the event included a review of instrument and event recorders, and interviews with plant operators to confirm plant conditions leading up to the event. The initial investigation identified that plant conditions immediately prior to the event were stable. There were no work activities or unusual operating evolutions in progress which had the potential for initiating the pump speed downshift logic. Manual initiation of the downshift and operator interaction were evaluated and found not to be a cause.

Troubleshooting of the Reactor Recirculation System Pump speed downshift instrument loops was initiated to identify component failures. Additionally, a formal root cause analysis was initiated to identify and evaluate possible root causes of the downshift. These investigations focused on evaluating the Reactor Recirculation Pump fast to slow speed logic circuitry. The following control interlocks IEL! are provided to initiate an automatic recirculation pump fast to slow speed downshift:

1. Low Recirculation Pump Suction/RPV Dome Steam Temperature Differential (less than 8.0 Degrees Fahrenheit).

2. End of Cycle Turbine Stop Valve Trip/Turbine Control Valve Fast Closure Trip.

3. Low Feedwater Flow (less than 3.43 million pounds-mass per hour).

4. Low Reactor Vessel Level (Level 3, 177.7 inches top of active fuel).

5. High Reactor Vessel Steam Dome Pressure (1083 psig).

Misfiring of optical isolator cards OB!, due to DC power supply BU! voltage surges, was initially considered a probable cause for this event, based on previous problems experienced. However, additional testing of the optical isolators showed they were not susceptible to misfiring. Based on this, voltage surges are considered a low probability for the cause of the event.

Component troubleshooting identified that the Reactor Recirculation Loop Inlet Temperature "B" Resistance Temperature Detector (RTD) TE! had failed. An intermittent failure mode of the "A" RTD was subsequently identified after additional testing was performed at simulated operating temperature conditions. This testing also included minor mechanical agitation of the RTD to simulate actual in

service conditions. The failure of both RTDs will satisfy the Low Recirculation Pump Suction/RPV Dome Steam Temperature Differential logic and initiate the Reactor Recirculation Pump fast to slow speed downshift.

The RTDs which failed were manufactured by Rosemount, Inc. (Model 187-1-5). While there have been several previous failures of these RTDs, no incident of concurrent failures of two RTDs to complete the downshift logic has occurred. A manufacturer's representative arrived on site to evaluate the test results and failure history of these RTDs. Several of the failed RTDs, including the two current failures, were returned to the manufacturer for failure analysis.

Based on the investigations completed, it was concluded that failure of both Reactor Recirculation Pump Suction Temperature A/B RTDs was the most probable cause of the event. Failure of both RTDs would satisfy the Low Recirculation Pump Suction/RPV Dome Steam Temperature Differential logic and initiate the Reactor Recirculation Pump fast to slow speed downshift. Other causes for downshift were evaluated and were determined to be of low probability. An independent assessment of the root cause of this event was also performed, which included personnel from other nuclear plants and General Electric. No other causes were identified as a result of this assessment.

IV. Corrective Action

Both failed Reactor Recirculation Pump Suction Temperature A/B RTDs have been replaced. A design modification was installed to provide control room annunciation if a failure of either RTD recurs. A troubleshooting contingency plan has been developed should further failures be identified. Several of the failed RTDs, including the two current failures, were returned to the manufacturer for failure analysis. Any further corrective actions will be determined pending the results of this analysis.

Additionally, a time events analyzer has been installed to monitor the Reactor Recirculation Pump fast to slow speed downshift circuitry should an unexpected downshift recur during the current Reactor Startup. A supplemental report will be submitted if any new information regarding the cause of this event is obtained.

Although voltage surges were not considered a cause for this event, a design modification was installed to provide filters in the optical isolator circuitry to suppress any potential voltage surges. Additional testing has been initiated to monitor the DC power supply for voltage surges.

Corrective actions for additional equipment malfunctions, described in Section II of this report, will be completed in accordance with site procedures for corrective action.

This event will be reviewed with licensed operators as part of routine operator requalification training.

V. Safety Analysis

The primary purpose of the Reactor Recirculation System is to provide forced circulation through the reactor core to achieve full power operation and permit variations in power level without control rod movement. Control interlocks are provided for the Reactor Recirculation Pumps to automatically downshift the pump from fast to slow speed. These controls are provided to prevent cavitation in Reactor Recirculation System components and mitigate the effects of various operational transients on reactor water level and reactivity.

Analysis of plant conditions at the time of the Reactor Recirculation pump fast to slow speed downshift confirmed that no real process parameters or transients required to initiate the above interlocks were present. With the reactor at full power, the pump speed downshift brought the reactor into the region of potential instability of the power to flow map. No evidence of a power oscillation or power level excursion was detected. Actual power during the event transient was calculated at 48 percent (1708 megawatts thermal). The reactor was manually scrammed to exit the region of potential instability of the power to flow map, and the plant shutdown was completed. Therefore, this event is not considered safety significant.

Energy Industry Identification System Codes are identified in the text as XX!.

ATTACHMENT 1 TO 9308110168 PAGE 1 OF 1

CENTERIOR ENERGY

PERRY NUCLEAR POWER PLANT Mail Address:
P.O. BOX 97 Robert A. Stratman
10 CENTER ROAD PERRY, OHIO 44081 VICE PRESIDENT-NUCLEAR
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(216) 259-3737

August 6, 1993
PY-CEI/NRR-1687 L

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 93-015

Dear Sir:

Enclosed is Licensee Event Report 93-015 for the Perry Nuclear Power Plant.

If you have any questions or require additional information, please contact Kevin Donovan, Manager - Licensing and Compliance at (216) 259-3737 extension 5606.

Sincerely,

Robert A. Stratman

RAS:AHL:ss

Enclosure: LER 93-015

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

Operating Companies
Cleveland Electric Illuminating
Toledo Edison

*** END OF DOCUMENT ***